Appendix A: Workscopes for U.S. University-led Program and/or Mission Supporting R&D Projects

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

INNOVATIVE NEW STRUCTURAL MATERIALS FOR MOLTEN CHLORIDE SALT FAST REACTOR APPLICATIONS (RC-1)

(FEDERAL POC – SUE LESICA & TECHNICAL POC – SAM SHAM)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Metallic structural components of Molten Salt Reactors (MSRs) have significant structural integrity challenges due to the extreme environments of high temperatures, corrosive coolants and neutron irradiation (including fission and transmutation products.) The selection of metallic alloys for structural applications is further complicated by the variety of MSR systems that are being considered, e.g., fast versus thermal spectrum reactor core, solid versus liquid fuel, and fluoride versus chloride salts. Existing ASME code qualified metallic alloys do not meet the challenges imposed by these extreme environments.

NEUP awards on developing the next generation MSR structural materials for applications in fluoride-based MSRs were made in FY18 and 19. The objective of this call is to develop high performance new metallic alloys that can be used for welded construction of structural components of molten chloride salt fast reactors using liquid fuel. Non-metallic materials are not within the scope of this call.

Characteristics of the new metallic alloy(s) to be considered include, but are not limited to, high temperature strength up to 900 to 950C, long-term thermal stability, chloride-based fuel salt compatibility, irradiation damage resistance, resistance to possible fission or transmutation product embrittlement, adequate tensile and creep ductility and weldability, all for the desired life times of the components. While not specifically a part of this call, the long-term goal of alloys developed under this effort would be their qualification for nuclear service under ASME Section III, Division 5, hence fabricability and potential capability for commercialization of any alloys developed are important. Innovative concepts, such as exploiting nano-scale interfaces within the alloy to trap defects and possible transmutation products, to address the material challenges are highly encouraged, as are novel applications of high-value experiments with integrated computation materials engineering for the development and testing of new metallic alloy(s).

While not required, interaction with molten chloride salt fast reactor developers on their system requirements is highly encouraged.

MICRO-REACTOR TECHNOLOGY DEVELOPMENT AND SUPPORT FOR DEPLOYMENT (RC-2)

The DOE's Micro-reactor Research, Development, and Demonstration (RD&D) Program supports technology development efforts for MW-class reactors with flexible deployment options. The program is seeking proposals that support the deployment case for micro-reactors, mature advanced technologies to reduce development time and cost, and/or validate modeling and simulation tools to enable their readiness for use in deployment design and licensing activities. These three areas are discussed in more detail below:

RC-2.1: MICRO-REACTOR DEPLOYMENT MARKETS

(FEDERAL POC – TOM SOWINSKI & TECHNICAL POC – JESS GEHIN)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

This work scope seeks proposals to analyze potential markets and applications that take advantage of unique micro-reactor characteristics. Micro-reactors are being designed to enable flexible siting that can support applications for remote areas (e.g., remote communities and mining operations) and for integration into microgrids in populated areas to provide resilient power as well as use for emergency operations related to natural disasters. Proposals are sought to perform techno-economic analyses on micro-reactor application topics such as:

- Requirements and ability to site micro-reactors in both remote and populated areas.

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- Assessment of micro-reactor needs for remote industries including mining, high value local data processing centers, and other applications unique to micro-reactors;
- Evaluation of micro-reactors use for increased resilience of the existing electrical grid and for microgrids.

Studies should incorporate current micro-reactor designs and concepts, avoid duplication of recent micro-reactor market studies, recommend micro-reactor design requirements and potential improvements relevant to specific applications, and provide a data-driven market assessment for a range of micro-reactor sizes.

RC-2.2: VALIDATION OF MICRO-REACTOR MODELING AND SIMULATION TOOLS
(FEDERAL POC – TOM SOWINSKI & TECHNICAL POC – JESS GEHIN)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The Micro-reactor RD&D program is seeking to enable the use of modeling and simulation capabilities being developed by the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program for use in micro-reactor design and licensing activities by supporting the validation of the tools. This includes performing validation of the tools with existing experimental data and performing new experiments that target validation needs. Specific areas for validation include:

- Neutronics validation specifically considering unique micro-reactor materials, such as high-temperature moderators
- Thermal/heat transfer for gas-cooled and heat-pipe-cooled micro-reactors including in-reactor heat transfer and coupling to heat exchangers
- Mechanics of structural systems such as solid-core block
- Integrated system performance validation

This work scope specifically is focused on validating NEAMS tools and maturing them for micro-reactor deployment applications (i.e., not on development of new modeling and simulation capabilities). Experimental work to develop validation data for current design concepts is encouraged. Proposals should connect the proposed experiments with a clear validation need.

RC-2.3: MICRO-REACTOR TECHNOLOGY DEVELOPMENT AND MATURATION
(FEDERAL POC – TOM SOWINSKI & TECHNICAL POC – JESS GEHIN)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Many micro-reactor concepts employ technologies developed and demonstrated in previous and existing DOE advanced reactor R&D programs. This work scope seeks the development of approaches to deploy innovative technologies and solutions specific to micro-reactors. These technologies and solutions should build upon and enhance (though not duplicate) previous/existing DOE advanced reactor technology RD&D efforts in the following areas:

- Reducing the amount of high-assay low-enriched uranium (HALEU) required for micro-reactor concepts.
- Accelerating manufacturing and fabrication approaches for unique micro-reactor components.
- Flexible siting options that reduce on-site preparation needs and supports site-independent designs.
- Transportation of fueled micro-reactors to site and return of the used micro-reactor.
- Micro-reactor operational and maintenance regimes that minimize staffing requirements.
- Integration of micro-reactor components such as innovative heat exchangers (heat pipe, gas, and/or liquid coolants) to power conversion systems and/or process heat systems.

Technologies and RD&D approaches should be described in detail and include a description of the expected

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improvements over the current state of the art and improvements in development schedule and cost. The technology readiness and development time line should also be described in detail.

RC-3: LIQUID METAL-COOLED FAST REACTOR TECHNOLOGY DEVELOPMENT AND DEMONSTRATION TO SUPPORT DEPLOYMENT
(FEDERAL POC – TOM SOWINSKI & TECHNICAL POC – CHRIS GRANDY)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The Department of Energy, National Laboratories, and U.S. nuclear industry are aggressively working to revive, revitalize, and expand U.S. nuclear energy capacity. Advanced non-light water reactors such as liquid metal-cooled fast reactor concepts offer the potential for significant improvements to safety, economics, and environmental performance to help sustain and expand the availability of nuclear power as a clean, reliable, and secure power source for our nation.

The Mechanisms Engineering Test Loop (METL) located at Argonne National Laboratory (ANL) is a sodium test facility that is designed to test small to intermediate-scale components and systems in order to develop advanced liquid metal technologies. Testing different components in METL is essential for the future of advanced liquid metal-cooled fast reactors as it provides invaluable performance data and reduces the risk of failures during plant operation. METL also provides developmental opportunities for the next generation of scientists, engineers, and designers who will ultimately lead the advancement of U.S. liquid metal technologies. The hands-on experience gained with METL will ultimately lead to better liquid metal technology programs that can support the commercialization of advanced liquid metal-cooled fast reactors.

This work scope seeks proposals to develop experiments, instrumentation, control strategies, and performance enhancing technologies for the METL facility that have the potential to subsequently be deployed and used by liquid metal (sodium or lead-cooled) fast reactor concepts proposed by U.S. nuclear industry. Experiments that offer the potential for significant overall benefits to reactor capital or operating cost reductions are of particular interest.

Examples of potentially beneficial METL experimental work and technology development areas include:

- *Development of test articles for testing in the Mechanisms Engineering Test Laboratory (METL) sodium loop facility* - The test articles should consider demonstration of innovative fast reactor sub-components (sensors, seals, mechanisms, etc.) or validation of key fast reactor behaviors under prototypic or near prototypic conditions
- *Advanced sensors and instrumentation* – Advanced liquid metal-cooled fast reactors will contain sensors and instrumentation that may be required to operate while immersed in the primary liquid metal coolant. These include: sensors for the rapid detection of hydrogen presence in sodium (which is indicative of a leak), the detection of impurities in the coolant (i.e., improvement of plugging meters or oxygen sensors), alternative methods of leak detection, improved sensors for level measurement, and other advanced sensors or instrumentation that improve the overall performance of the liquid metal-cooled fast reactor systems
- *Thermal hydraulic testing in prototypic sodium environment* – A thermal hydraulic test loop could be used to acquire distributed temperature data in the cold and hot pools of a small scale sodium-cooled fast reactor during simulated nominal and protected/unprotected loss of flow accidents. This testing could allow for the articulation of the heated region in the core for use in a parametric study of intermediate heat exchanger (IHX)/core outlet height difference and its effect on thermal stratification of sodium in the hot pool. Ultimately, this data would be used for validating CFD and systems level code
- *Mechanisms for self-actuated control and shutdown systems* – Components have been conceived by various designers to provide added defense-in-depth for reducing the consequences of beyond-design-basis accidents. These self-actuated control and shutdown mechanisms include devices such as curie-point magnets and fusible linkages. Performance of such mechanisms in a liquid metal environment may

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further enhance the safety case of liquid metal-cooled reactors
- *In-service repair technologies* – These systems include visualization sensors for immersed coolant applications and technologies for the welding and repair of structures in contact with the primary coolant
- *Performance improvement technologies for METL* – Technologies for improving the performance of liquid metal test loops including rugged high temperature resistance heating systems, improved insulation technology, improved sodium leak detection and identification technologies, vessel support technologies that reduce heat losses, improved clamp on flow meters, thermal monitoring, etc.
- *Health Monitoring of METL systems and components* - Development of deployable sensors and prognostic techniques for demonstration in METL that can be used to monitor and quantify materials degradation in liquid metal-cooled fast reactor primary systems. Of interest are technologies that are able to detect degradation early, can survive in typical liquid metal-cooled fast reactor environments over extended periods of time, and can be embedded in/on structural materials to enable structural health monitoring (e.g., nondestructive examination techniques, remote or automated inspection techniques including visualization in optically opaque coolants). Consideration should be given to deployment issues that may arise, such as powering the sensor and data exfiltration needs

Though proposals are not limited to the example work areas above, applicants should indicate how their proposed work will support testing articles in the METL facility to develop and deploy technologies for use in U.S. liquid metal-cooled fast reactor concepts and/or to increase the performance of the METL facility to support current DOE, national laboratory, and U.S. nuclear industry liquid metal-cooled fast reactor deployment and commercialization R&D initiatives.

The following web site contains more information on METL:

https://www.anl.gov/nse/mechanisms-engineering-test-loop-facility

HTGR TRISO FUEL PARTICLE MATERIALS (RC-4)

RC-4.1: TRISO FUEL BUFFER LAYER BEHAVIOR DURING NEUTRON IRRADIATION (FEDERAL POC – DIANA LI & TECHNICAL POC – PAUL DEMKOWICZ) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $800,000)

The AGR TRISO program has demonstrated very low coating failure fractions for tristructural-isotropic (TRISO) coated particle fuel under irradiation and post-irradiation safety testing at 1600–1800°C. A common failure mechanism observed in the few AGR particles that exhibited elevated release of fission products through degraded SiC layers was associated with how the buffer layer behaved during irradiation. In these particles, dimensional changes in the buffer led to cracks in the inner pyrolytic carbon (IPyC) that exposed the SiC to higher concentrations of reactive fission products (primarily palladium in UCO-TRISO and CO in UO2-TRISO) [1,2]. The buffer is a heterogenous, carbonaceous material with significant porosity (~50% dense). Radiation-induced densification of the buffer results in significant dimensional change not observed in the denser pyrocarbon layers. Post-irradiation examination has shown that over certain temperature and fluence ranges, shrinkage of the buffer layer is usually accompanied by fragmenting or tearing of the buffer material.

Observations of irradiated AGR TRISO particles [3,4] suggest that the preferred buffer evolution is when stress from radiation-induced densification of the buffer results in circumferential tearing in the buffer close to the buffer/inner pyrolytic carbon (IPyC) interface, such that the majority of the buffer layer is decoupled from the IPyC. This allows the thick inner portion of the buffer to freely shrink down around the kernel, and stress on the IPyC layer is reduced because only a thin outer portion of the buffer remains attached. In contrast, IPyC fracture has been observed to be associated with locations where the buffer and IPyC layers were connected and a developing buffer fracture intersected the buffer/IPyC interface [1]. Circumferential bands of varying density/porosity are observed in the as-fabricated buffer as a result of variability during coating as particles cycle through the fluidized bed and the rate of surface area-normalized buffer material deposition changes with

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increase of particle diameter. This banding may impact how and where buffer tearing occurs and may offer strategies for encouraging desired behavior under irradiation.

Research proposals are sought that focus on the time-dependent evolution of the buffer and IPyC dimensional and stress states as a function of particle dimension, initial buffer microstructure, irradiation temperature, neutron flux, and neutron fluence. Research proposals are sought that specifically focus on investigating these specific TRISO coating layer properties: (a) Buffer: elastic modulus, tensile strength; (b) Pyrocarbon (IPyC and OPyC): elastic modulus, tensile strength, irradiation-induced creep and dimensional change; and (c) Buffer-IPyC bond strength, and developing computational models to describe the buffer and IPyC phenomena. Of particular interest are the radiation induced shrinkage of the buffer and associated stress state leading to internal buffer tearing, and the analysis should be focused on length scales associated with the internal buffer tearing phenomena, which is on the order of tens of microns. It would also be advantageous for computational models to consider the stresses induced on the IPyC layer due to these buffer behaviors, as these can contribute (along with other factors such as irradiation-induced strain in the IPyC) to IPyC fracture and separation from the SiC layer. The impact of kernel swelling should also be included as another source of stress in the buffer that can lead to dimensional changes and buffer fracture. In many cases, material properties are not known (e.g., buffer strength, buffer elastic modulus, and buffer-IPyC bond strength). Parametric studies over a range of material properties may be used to compare modeling results to available data from observation of TRISO fuel irradiated by the AGR program.

All project tasks must be performed to NQA-1 standards. Data, experiments, fuel performance computational modelling information, and any calculations shall be submitted to the Idaho National Laboratory's NGNP Data Management and Analysis System (NDMAS).

[1] J.D. Hunn et al., "Detection and analysis of particles with failed SiC in AGR-1 fuel compacts," Nucl. Eng. Des. 306 (2016) 36–46.

[2] R.N. Morris et al., "Initial results from safety testing of US AGR-2 irradiation test fuel," Nucl. Eng. Des. 329 (2018) 124–133.

[3] F.J. Rice et al., "Ceramography of irradiated TRISO fuel from the AGR-2 experiment, " INL/EXT-16-39462, Rev. 0, 2016.

[4] J.D. Hunn et al., "Initial examination of fuel compacts and TRISO particles from the US AGR-2 irradiation test," Nucl. Eng. Des. 329 (2018) 89–101.

RC-4.2: ROBUST INDIVIDUAL TRISO-FUELED PEBBLE IDENTIFICATION METHOD FOR EX-CORE EVALUATION
(FEDERAL POC – DIANA LI & TECHNICAL POC – PAUL DEMKOWICZ)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Several advanced gas-cooled and salt-cooled reactor designs currently under consideration utilize spherical fuel pebbles containing thousands of TRISO fuel particles. These pebbles are not static, but move through the reactor core and are periodically removed in order to measure the burnup nondestructively [1]. Pebbles that are below a specified target burnup value are returned to the core, while pebbles that exceed this target are sent to spent fuel storage. New pebbles are added to the core to replace those that were removed to spent fuel storage. In pebble-bed reactors that have been operated to date, the identity of the individual pebbles in the core (typically many thousands of pebbles, depending on design) have not been tracked during operation. However, there are advantages to maintaining pebble identity, including determination of pebble transit time to validate computational models [2]. This can help determine if any pebbles are retained for unexpectedly long times in the core which could result in excessive burnup accumulation. Therefore it may be useful to tag individual pebbles in order to track the identities as they cycle ***out*** of the reactor core ***following irradiation*** for analysis and potential core reentry. However, pebble identification poses several inherent challenges including: (a) potential abrasion

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or degradation of the pebble surface, (b) the high-temperature neutron irradiation environment, and (c) the large number of pebbles that need to be tracked and catalogued during reactor operations, which requires a relatively rapid ex-core burnup assessment [1] period following in-core irradiation. The development of a novel method which overcomes these challenges and provides rapid, reliable, and robust pebble identification would be of great interest and use for several pebble-based advanced reactor designs.

The proposed "tagging and tracking" identification method needs to be very robust and radiation resistant, and should encompass a realistic range of HTGR normal operation and accident temperatures and neutron fluence levels: (1) burnup to 20% fissions per initial metal atom (FIMA), (2) fast neutron fluences up to 4.5E25 n/m^2 (E > 0.18 MeV), and (3) normal operational irradiation temperatures ranging from approximately 900 to 1250°C. The proposed identification technique may use existing gamma spectroscopy and passive neutron counting of spontaneous fission neutrons used in proposed pebble burnup measurement systems [1] and/or additional detection methods to read the individual pebble tag identification. Proposals for developing burnup monitoring systems are not solicited in this call.

All work (e.g., experiments and calculations) must be performed to NQA-1 standards. Data, experiments, and any calculations shall be submitted to the Idaho National Laboratory's NGNP Data Management and Analysis System (NDMAS).

1. Su, Bingjing, "Design and Construction of a Prototype Advanced On-Line Fuel Burn-up Monitoring System for the Modular Pebble Bed Reactor," University of Cincinnati, NERI Project 00-100, Final Report, DE-FG07-00SF22172, March 30, 2004, available at: https://www.osti.gov/servlets/purl/822595.

2. Chen, Hao, et al., "Quantitative analysis of uncertainty from pebble flow in HTR," Nucl. Eng. Des. 295 (2015) 338-345.

RC-5: EXPERIMENTAL VALIDATION OF HIGH TEMPERATURE GAS REACTOR (HTGR) SIMULATIONS
(FEDERAL POC – DIANA LI & TECHNICAL POC – GERHARD STRYDOM)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Experimental validation of High Temperature Gas Reactor (HTGR) simulations is focused on providing data of high temperature gas-cooled reactor (prismatic or pebble bed) phenomena for the validation of system and computational fluid dynamics models. These phenomena are relevant to core safety and performance. Two scenarios of interest to core designers and safety analysts involve disruptions of the nominal helium flow rate and transition to either the Pressurized or Depressurized Conduction Cooldown (PCC/DCC) events [1], [2], [3]. The PCC is initiated by a helium circulator or turbine trip, while the DCC is the result of a slow depressurization after a leak or break in the primary pressure boundary. Most HTGR reactor systems incorporate water-or air-cooled Reactor Cavity Cooling System (RCCS) [2] as a final heat sink and for protection of the vessel and concrete confinement structures.

This work scope entails the evaluation of two elements - degraded or asymmetric RCCS performance and a plenum-to-plenum natural circulation characterization. Proposals utilizing existing integral and/or separate effects test facilities (with modifications) are encouraged, in addition to proposals suggesting development of new facilities.

In previous studies (e.g. reference [2]), the RCCS performance during PCC/DCC events were mostly characterized in separate effects or integral facilities assuming as-designed operation. In reality, various failure modes of the RCCS can result in either degraded (i.e. at a lower total heat removal rate) or asymmetric (i.e. an azimuthal section of the RCCS is non-operational) performance. The impact of these conditions on localized vessel and concrete temperatures should be characterized. In addition to the PCC event (i.e. terminating forced cooling and remaining at a pressurized condition), it is also desired that three classes of break events be included

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in the "compromised" RCCS test matrix: slow (depressurization over 12 hours), medium (1-6 hours) and fast (< 10 minutes) to assess the effects of blowdown periods on the figures of merit. If the test facility geometry allows different break locations (e.g. top and bottom breaks and inlet/outlet plena), this would enable various natural convection regimes to establish over different time-scales.

Finally, Principal Investigators are also invited to propose an experimental characterization of low-velocity (natural circulation) plenum-to-plenum gas flow at prototypical conditions. A flow reversal condition can for example occur during a PCC, i.e. hot gas flows upwards from the outlet plenum to the inlet plenum via the control rod and other bypass flow channels This facility could be part of a larger (new or existing) experimental setup, or a dedicated facility that would allow assessment of the natural circulation flow between two large void regions.

Principal Investigators are encouraged to consult with US-based HTGR vendors (Framatome, X-Energy, etc.) to refine the experiment design and test matrix. A literature review of previous experimental work performed (e.g. [2]) and the HTGR community V&V needs [4], specifically for CFD codes, would be expected to leverage previous recommendations and lessons-learned.

All experiments must be performed to NQA-1 standards. Data, experiments, and calculations shall be submitted to the Idaho National Laboratory's NGNP Data Management and Analysis System (NDMAS). Assistance shall be provided by Idaho National Laboratory for NDMAS use and ensuring NQA-1 standards are properly established.

References:

[1] Sterbentz, J.W., et al,. (2016). High-Temperature Gas-Cooled Test Reactor Point Design. INL/EXT-16-38296, Idaho National Laboratory.

[2] Yang, Se Ro & Kappes, Ethan & Nguyen, Thien & Vaghetto, Rodolfo & Hassan, Y. (2018). Experimental study on 1/28 scaled NGNP HTGR reactor building test facility response to depressurization event. Annals of Nuclear Energy. 114. 154-164. 10.1016/j.anucene.2017.12.023.

[3] NGNP Deliverable to DOE: Project Recommendations – Evaluation and Testing of HTGR Reactor Building Response to Depressurization Accidents, AREVA Document 12-9273530-00, July 2017.

[4] Schultz, R.R., et al., (2017). Identification and Characterization of Thermal Fluid Phenomena Associated with Selected Operating/Accident Scenarios in Modular High Temperature Gas-cooled Reactors. INL/EXT-17-34218, Idaho National Laboratory.

RC-6: FLUORIDE SALT-COOLED HIGH TEMPERATURE REACTOR (FHR)

The Fluoride Salt-cooled High Temperature Reactor (FHR) focus area is seeking to address either one of the two areas discussed below.

RC-6.1: OPTIMIZED FLUORIDE SALT PIPE JOINTS FOR FLUORIDE SALT COOLED HIGH TEMPERATURE REACTORS
(FEDERAL POC – DIANA LI & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Flanges that require repeated sealing along with thermal cycling for service conditions above 500 °C with low internal pressures remain a substantial engineering challenge. Bolts creep and relax, threads gall, and the torque used to seal at room temperature is inappropriate at operating temperature. Differential thermal expansion of multi material systems frequently result in leaks. Resilient metal seals require precise surfaces to achieve/maintain seal. Other designs rely upon internal pressure to force a compressible seal into a mated backing. The historic MSR program relied upon frozen salt gaskets which were not testable prior to filling along

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with high-temperature bolting. All of the available designs are difficult to make-up and remove using remote/automated tooling. Optimized fluoride salt pipe joints are sought that are suitable for both large and small pipes, can be repeatedly joined and disconnected, can be tested prior to filling with salt, function when subjected to repeated thermal cycling, do not require internal pressure for sealing, and are tolerant of common engineering tolerances (roughness and alignment) of mating surfaces.

Due to the relatively early stage of maturity of FHR facilities and significant resources required, establishing an NQA-1 program may not be feasible, however priority will be given to experiments that are performed to NQA-1 standards.

RC-6.2: PUMP SCALING TECHNOLOGY FOR FLUORIDE SALT COOLED HIGH TEMPERATURE REACTORS

(FEDERAL POC – DIANA LI & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The pump shaft seals and oil lubricated shaft bearings were among the most problematic technologies at the MSRE with hydrocarbon lubricant leaking into the fuel salt with a wide range of undesirable impacts. Also, as FHR technology progresses from laboratory to industrial scale, its component technologies need to scale. Long-shafted, cantilever pumps are the currently leading candidate for larger-scale FHR primary pumps. Gas-foil seals (to avoid hydrocarbon lubricants), however, require precise, well aligned shaft and seal surfaces, which, are more challenging at scale and, in practice, require bearings within the salt. Fluoride salt compatible bearings remain largely unproven. Larger gas gaps leave larger paths for gaseous radionuclide releases. Development and demonstration of scalable fluoride salt pump component technologies (seals and bearings) are requested.

Due to the relatively early stage of maturity of FHR facilities and significant resources required, establishing an NQA-1 program may not be feasible, however priority will be given to experiments that are performed to NQA-1 standards.

RC-7: PLANT MODERNIZATION R&D PATHWAY

RC-7.1: DIGITAL INSTRUMENTATION AND CONTROL

(FEDERAL POC – ALISON HAHN & TECHNICAL POC – CRAIG PRIMER)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

One of the current challenges of digital modernization for operating Light Water Reactors is digital conversion, enabling elimination of analog field circuits in the plant I&C systems, including the field logic devices (relays, timers, etc.) and associated cables. Developing the digital system software equivalent of an analog circuit is labor-intensive and susceptible to error. An automated or semi-automated means of converting analog circuit design information (derived from electronic design drawings) to equivalent digital system function block logic (or other types digital logic) would be a significant enabler of full digital modernization for nuclear plants. Such a capability should generate code that can be automatically tested either in the target digital platform or a simulation for correct operation.

Proposed research will address:

- Develop an automated or semi-automated means of converting analog circuit design information (derived from electronic design drawings) to equivalent digital system function block logic (or other types of digital logic). This capability will further enable the function block logic (or other digital logic) to be automatically

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tested for correct operation either on the target digital platform or an equivalent simulator.

RC-7.2: VIRTUALIZED DISTRIBUTED CONTROL SYSTEMS FOR NUCLEAR POWER PLANTS
(FEDERAL POC – ALISON HAHN & TECHNICAL POC – CRAIG PRIMER)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Harvesting and reuse of process control application software and use of IT-based hardware at basic control level (Purdue Model Level 1) within a distributed control system (DCS) has the potential to dramatically reduce technology refresh costs and provide a long term, sustainable lifecycle support strategy to address digital obsolescence. Up to this point, efforts by industrial control system suppliers have been largely constrained to Level 2 and higher within the Purdue Model. Virtualized software (and its imbedded intellectual property) can be migrated to new IT based hardware when needed, reducing the cost of performing technology refreshes at Level 2 and above. Research is needed to determine the viability of virtualizing the function of DCS process controllers at Level 1. This would largely decouple core DCS control application software from the custom built and closely intertwined hardware and software of current DCS industry product. Controller application software could then be migrated to a new IT based hardware.

Proposed research will address:

- Develop methods and techniques to allow virtualization of Distributed Control System (DCS) Purdue Model Level 1 hardware & software while maintaining key properties of current DCSs such as control segmentation, low latency, determinism, redundancy, fault tolerance, and graceful degradation.
    - Identify logical and physical architecture attributes/properties that enable virtualization of DCS Purdue Model Level 1 above the input/output interface to physical processes being monitored and controlled.
    - Demonstrate the proposed architecture ensuring current attributes (e.g. control segmentation, determinism, redundancy, fault tolerance, graceful degradation) of current DCSs at Purdue Model Level 1 are maintained.

RC-7.3: REDUCING HUMAN FACTOR UNCERTAINTY USING ARTIFICIAL INTELLIGENCE IN OPERATION AND MAINTENANCE OF NUCLEAR POWER PLANTS
(FEDERAL POC – ALISON HAHN & TECHNICAL POC – CRAIG PRIMER)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The labor-centric business model of the existing domestic fleet of nuclear power plants is presenting an economical sustainability challenge. Today, operation and maintenance (O&M) involve several activities such as testing, inspection, data collection, interpretation, and decisions-making, that are performed by large number of skilled field workers. This introduces potential for human factors errors and variability between activities that are at times difficult to correlate due to missing record or sparseness.

Proposed research will:

- Leverage advancement in technologies to automate possible O&M activities along with explainable artificial intelligence and machine learning research to analyze all the data sources. Innovative and scientifically strong proposals in the area of artificial intelligence and machine learning techniques to reduce human error, variability, and uncertainty in data interpretation and decision-making. The outcomes of the research are expected to provide consistent interpretation of data, diagnosis of any potential problem, estimate future state, make acceptable recommendation, and quantify the uncertainty in decision-making. The demonstration

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of the developed concept in a representative environment is preferred to enable implementation of the developed concept in a nuclear power plant.

RC-8: RISK-INFORMED SYSTEMS ANALYSIS R&D PATHWAY

RC-8: EVALUATION OF PHYSICAL PHENOMENA DATA IMPACT AND IMPROVEMENTS
(FEDERAL POC – ALISON HAHN & TECHNICAL POC – CURTIS SMITH)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Modeling the effects of physical events (fire, flood, high winds, etc.) in a nuclear power plant's risk assessment involves using data and a standard of practice guidance. This data and guidance typically use a conservative evaluation based on historical data, expert judgement, and experiments of the physical phenomena. Further, some of the historical experiments used in analysis today were performed before the availability of advanced measurement tools and controlled environments. These unknowns in input parameters and lack of resolution in results causes uncertainties and conservative decisions in order to cover knowledge gaps.

Some small changes in results or reduction in uncertainty can have a large effect in results of a PRA model, such as the heat release rate curve in a fire model. However, recreating all these experiments may be cost prohibitive and often unnecessary. We request a project to research and evaluate the data used for hazard guidance in phenomena-driven areas and develop a method to determine significant contributors to uncertainty and determining what rerun or new experiments would be of most value (i.e., change in data/guidance leading to a difference in risk compared to the cost of new experiments). Possible work could include experiment determined to be of significant value for a specific phenomenon.

RC-9: MATERIALS RESEARCH PATHWAY

RC-9: ELUCIDATING HOW WATER CHEMISTRY AFFECTS THE CORROSION SENSITIVITY OF PRISTINE STAINLESS STEEL IN NUCLEAR POWER PLANTS
(FEDERAL POC – ALISON HAHN & TECHNICAL POC – XIANG (FRANK) CHEN)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Reductions in the operating costs of nuclear power plants (NPPs) are mandatory to pass the benefits of cost-effective electric power generation to the end-consumer. One possibility of operating cost reduction involves switching the alkalinization agent from LiOH to KOH. While a cost-effective possibility, the impacts of such a change and the consequent water chemistry alterations on the corrosion processes and NPP core-internal component service-life remain poorly understood; more so in the case of irradiated materials (e.g., stainless steels).

Therefore, proposals are sought to develop a robust new understanding of the mechanisms and processes by which ions interact with alloy surfaces bearing diverse microstructures, in aqueous solution, and affect the near-surface local electrochemical environment thereby resulting in the onset of corrosion processes - in pristine stainless steels, and those subjected to irradiation. It is critical to understand how water chemistry changes (e.g., from LiOH to KOH) may affect the degradation and durability of NPP core-internals; exploring simultaneously how contributions of mechanical stress factors affect corrosion sensitivity will be considered a key advantage. The systematic integration of pioneering experimental analyses with integrated computational materials engineering (ICME) for developing new fundamental insights is highly encouraged.

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

RC-10: ADVANCED SMALL MODULAR REACTOR R&D (FEDERAL POC – TIM BEVILLE & TECHNICAL POC – TBD)

The DOE's Advanced SMR Research and Development (R&D) Program supports technology development efforts for domestic SMR designs that can provide safe, affordable and resilient power generation options to meet the nation's economic, energy security and environmental goals. SMRs are nuclear power plants that are smaller in size (approximately 50 to 300 megawatts electric) than current generation base load plants (many are 1,000 megawatts electric or higher). These smaller, compact designs consist of major components and modules that can be factory-fabricated and transported to a nuclear power site by truck, rail, or barge. The Department is currently working with industry, the national laboratories and academia to advance the certification, licensing, and siting of domestic SMR designs, and to reduce economic, technical, and regulatory barriers to their deployment. DOE's work is focused on domestic deployment of SMRs, but many of our international partners are also interested in SMR technology and its maturation. This effort under the NEUP is seeking applications that can develop technologies to support the accelerated development and deployment of SMR designs, and to conduct economic and market analysis that can provide insights into the potential for reducing the costs as well as the domestic and international market penetration of SMRs. These areas are discussed in more detail below:

RC-10.1: TECHNOLOGIES TO SUPPORT SMR DEVELOPMENT AND COMMERCIALIZATION (FEDERAL POC – TIM BEVILLE & TECHNICAL POC – TBD) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $800,000)

This work scope seeks applications that would develop technologies, capabilities and methodologies specific to SMR characteristics and environments that would help to improve the deployment potential of SMRs. Applications should support a broad range of SMR technologies (i.e., light water, gas, liquid metal and molten salt cooled designs), and offer specific safety, safeguards, operational, and economic efficiency improvements for this class of reactor designs. Applicants should focus on areas that address the niche characteristics of SMRs, such as the simplified designs, operational flexibility, multi-unit deployment, potential for fleet-level deployment, potential for below grade siting, and other key aspects. Examples of technology development areas where applications are sought include, but are not limited to, the following:

- Remote or autonomous operation capabilities;
- Control room improvements;
- Compact, high efficiency heat exchanger and steam generator designs;
- Improved penetration technologies for primary system components;
- On-line power monitoring capabilities for anticipated advanced SMR cores;
- Methodologies or use of robotics for remote or automated maintenance in confined environments and high radiation fields; and,
- Ability to conduct material accountability for advanced fuel designs in fleet-level deployment.

RC-10.2: SMR MARKET ANALYSES (FEDERAL POC – TIM BEVILLE & TECHNICAL POC – TBD) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $800,000)

This work scope seeks applications to analyze potential markets and applications that take advantage of SMR characteristics. SMRs are being designed to enable siting at areas that are not feasible for the current generation of large light water reactors but require more power than can be provided by micro-reactor systems. This includes applications to replace retiring fossil plants where deployment of a large LWR is not economically feasible, for powering remote areas (e.g., remote communities, mission-critical installations, and mining operations), and to potentially provide a generation source for microgrids in populated areas to provide resilient

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

power. Applications are sought to perform analyses on topics such as, but not limited to, the following:

- Feasibility of siting SMRs in both remote and populated areas;
- Examination of markets that are specifically trying to retire coal plants that are in the same output range as known SMR designs, and could be "sweet spots" for SMR deployment;
- Assessment of SMR capabilities to support remote industries such as mining operations, remote population centers, to provide resilient power to permanent, mission-critical facilities, and other potential applications;
- Evaluation of SMR capabilities to provide increased resilience of the existing electrical grid and for possible microgrid applications.

Studies should incorporate information consistent with the capabilities of current domestic SMR designs and concepts of any technology type, should avoid duplication of existing SMR market studies, recommend SMR design requirements and potential improvements relevant to specific applications, and provide a data-driven market assessment for a range of SMR sizes and performance parameters.

RC-10.3: SMR ECONOMIC ANALYSES
(FEDERAL POC – TIM BEVILLE & TECHNICAL POC – TBD)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

This work scope seeks applications to conduct analyses or sensitivity studies that can address economic impacts of various SMR plant design, fabrication, construction, operation, and decommissioning activities, state and national-level energy policies and initiatives, and financing structures on the levelized cost of electricity (LCOE) of SMRs. Applicants should focus on areas that address the niche economic characteristics of SMRs, such as potential for factory fabrication, ability to transport major components and modules, ability to replace fossil generation, fleet deployment aspects, and other key elements. The goal of the studies should be to bolster and improve the business case for domestic deployment of SMRs. Applications may address the impact on economics of a wide range of SMR-specific factors including, but not limited to, the following:

- Fabrication and construction improvements;
- Supply chain development and improvement activities;
- Unique transportation methodologies for major SMR components and modules that could improve construction schedules;
- Unique considerations for fleet-level fuel transportation and disposition;
- Decommissioning policies and methodologies;
- Financing structures that would support incremental deployment of multiple plant modules, or plant deployments at multiple locations; and,
- Potential federal and state policy changes, including actions such as changes in tax structures and valuation of resilience capabilities.

Studies should incorporate information consistent with the capabilities of current domestic SMR designs and concepts of any technology type, should avoid duplication of existing SMR economic studies, and recommend potential improvements relevant to areas assessed.

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

MATERIAL RECOVERY AND WASTE FORM DEVELOPMENT (FC-1)

This program element develops innovative methods to separate reusable fractions of used nuclear fuel (UNF) and manage the resulting wastes. The program employs a long-term, science-based approach to foster innovative and transformational technology solutions and applies the unique nuclear fuel cycle chemistry expertise and technical capabilities to a broad range of civil nuclear energy applications. These chemical technologies, when combined with advanced reactors and their fuels, form the basis of advanced fuel cycles for sustainable and potentially growing nuclear power in the U.S.

FC-1.1: NUCLEAR FUEL CYCLE CHEMISTRY
(FEDERAL POC – STEPHEN KUNG & TECHNICAL POC – TERRY TODD)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $600,000)

Complexation Chemistry, Actinides Chemistry, Radiation Induced Degradation and Chemistry

A fundamental knowledge of the electronic structure of the complexes of actinides will expand our ability to predict their behavior quantitatively under conditions relevant to all stages in fuel recycling. To better controlling and tailoring chemical species and properties of actinides and fission products will enhanced separation and recovery efficiencies. There is also a need for an improved understanding of the fundamental processes that affect the formation of radicals and ultimately control the accumulation of radiation-induced damage to separation systems. For example, gamma radiation is known to produce radicals that can affect the oxidation states of multivalent cations in solution. Fundamental knowledge of the chemical speciation and partitioning of multivalent cations (e.g. Np, Tc, Zr) in advanced extraction process under high irradiation fields will improve processes being studied. Proposals are requested to enhance fundamental understanding of fuel cycle chemistry and to develop novel new approaches for advanced reactor used fuels treatments and recycling.

FC-1.2: ELECTROCHEMICAL SEPARATIONS
(FEDERAL POC – STEPHEN KUNG & TECHNICAL POC –MARK WILLIAMSON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 2 YEARS AND $400,000)

The solubility limit of fission products in molten chloride salts becomes one limit that impacts the timing for salt treatment and recycle, or disposal. Solubility data as a function of temperature and composition is available for many binary and ternary chloride systems especially those that contain alkali and alkaline earth chlorides. However, similar data is lacking for complex multicomponent chloride salt systems especially those involving the lanthanide and actinide elements. Proposals are requested to establish fundamental thermochemical data in chloride salt systems with an emphasis on multicomponent solubility of the fission product chlorides. Experimental work may be supplemented by thermodynamic data simulation to yield a more complete understanding and predictive models of the solid-liquid equilibria in the complex chloride systems.

FC-1.3A: WASTE FORMS DEVELOPMENT AND OFF-GAS CAPTURE
(FEDERAL POC – KIMBERLY GRAY & TECHNICAL POC – BOB JUBIN)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 2 YEARS AND $400,000)

IODINE IMMOBILIZATION FROM CAUSTIC SCRUBBER SOLUTION- The size and cost of using silver-based solid sorbent material alone to remove I-129 are relatively large. Recent studies have demonstrated that the combination of a caustic scrubber followed by a silver-based solid sorbent polishing bed may reduce the sorbent usage by ~90%. The recovered iodine, associated tramp halogens, and co-absorbed carbon (C-12 plus C-14) in the caustic scrubber solution must be converted to a highly stable and corrosion resistant waste form. Proposals are requested

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to develop conversion process(es) for the spent scrubber bottoms solution into highly durable waste form. The proposed waste forms/processes that avoid the use of silver or other hazardous metals are preferred. The conversion process should also consider, and experimentally verify, the fraction of the halogens and carbon that would be released during the conversion process and that would require recapture. The proposed effort should include the production of multiple, 20 gram monolithic waste form test samples that would be provided to the DOE National Laboratories for testing beginning no later than 15 months into the effort and continuing to the conclusion of the proposed effort. Samples of the proposed waste forms would be evaluated using the facilities and methods developed within the DOE National Laboratory complex.

FC-1.3B: WASTE FORMS DEVELOPMENT AND OFF-GAS CAPTURE
(FEDERAL POC – KIMBERLY GRAY & TECHNICAL POC –JOHN VIENNA)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 2 YEARS AND $400,000)

SALT WASTE RECYCLE AND IMMOBILIZATION– Some advanced reactors and fuel cycles generate alkali- and alkaline-earth halide-based salt waste streams. These streams contain fission products and some actinides and must be treated to immobilize the radioactive components. The national laboratories are developing a method to dehalogenate the salt waste using phosphate compounds. This process generates a phosphate liquid containing fission products and actinides. Durable phosphate-based waste glasses may be used to immobilize the resulting molten phosphate-based streams. Proposals are requested to develop a highly durable and easily processable phosphate-based waste glass. The waste form should be developed to be processed efficiently in existing waste glass melter technologies and result in acceptable glass when slow cooled in full scale waste glass canisters. The proposed effort should include the production of multiple, 20 gram monolithic waste form test samples that would be provided to the DOE National Laboratories for testing beginning no later than 15 months into the effort and continuing to the conclusion of the proposed effort. Samples of the proposed waste forms would be evaluated using the facilities and methods developed within the DOE National Laboratory complex.

ADVANCED FUELS (FC-2)

FC-2.1: NDE TECHNIQUES FOR ASSESSING INTEGRITY OF COATED CLADDING TUBES
(FEDERAL POC – FRANK GOLDNER & TECHNICAL POC – TARIK SALEH)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Near-term Accident Tolerant Fuel (ATF) technologies currently under development include coated cladding concepts, including coatings on zirconium alloys and SiC-SiC composites. Coating thicknesses under development by commercial fuel vendors are typically between 2-20 µm in thickness. In order to enable and support cost effective manufacturing, non-destructive quality assurance techniques are needed to: 1) verify the thickness and uniformity of the coating layers, 2) identify areas of missing coating layers, and 3) confirm coating-substrate bond quality.

The present call seeks to stimulate proposals to develop and demonstrate innovative, non-destructive, and potentially high-throughput inspection techniques for coated zirconium alloy and/or coated SiC-SiC cladding tubes that address one or more of these characterization needs. It is expected that applicants will need to work in collaboration with one or more of the DOE-sponsored ATF vendors (i.e., Westinghouse, General Electric, and Framatome) in order to obtain coated cladding specimens for testing. Applicants should clearly identify the coating-cladding combination to be investigated and which fuel vender will supply the applicant with specimens for testing. The applicant and the fuel vendor are responsible for establishing the intellectual property protection plan for their collaboration and should provide a written confirmation of this established framework in the final proposal. No portion of the funds in this award area may be used to develop new coating technologies; the focus must be on development of inspection techniques in support of commercial concepts.

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FC-2.2: INVESTIGATIONS OF CARBIDE AND NITRIDE FUEL SYSTEMS FOR ADVANCED FAST REACTORS
(FEDERAL POC – JANELLE EDDINS & TECHNICAL POC –ANDY NELSON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

A wide range of solid fuel advanced fast reactors are presently under investigation for numerous applications ranging beyond historic transmutation missions to include once-through fuel cycle concepts. Uranium-zirconium (U-Zr) metallic fuels and mixed oxides (MOX) have been the reference fuels for the majority of historic fast reactor concepts beyond commercial power generation. Both U-Zr and MOX fuel forms have myriad benefits, but both possess well known limitations as nuclear fuels for certain applications. Alternate fast reactor fuels have been proposed and assessed to varying degrees over previous decades, but none have been advanced to the degree of either U-Zr or MOX. There is renewed interest among commercial advanced reactor designers for high density ceramic fuels such as uranium monocarbide (UC) and uranium mononitride (UN) as candidates for gas-cooled or lead-cooled fast reactors. In comparison to metallic and oxide fuels, these fuel forms have seen much less assessment and evaluation. The more limited attention given to carbides and nitrides as fast reactor fuels has limited the use of modern experimental and simulation tools to better understand the capabilities, uncertainties, and limitations in service of these fuels.

The present call intends to stimulate proposals that address aspects of carbide and nitride fuels for advanced reactors through leveraging of modern experimental methods and capabilities as well as modeling and simulation tools. Focus should be restricted to previously-defined solid fuel fast reactor concepts. The goal is to contribute to advancing the technological readiness levels of UC and UN fuel forms as applied to gas fast reactor and lead fast reactor systems, respectively, in anticipation of future commercial use. It is anticipated, but not required, that both experimental and modern modeling and simulation tools will play important roles in a successful proposal. Focus may be placed on either a major challenge of a single component of the fuel or extended to include the fuel/cladding/coolant system. Specifically, proposals that address high priority items related to these fuel types such as chemical compatibility between UN and Pb coolant, or compatibility of ferritic-martensitic steels or Al-forming steels with Pb coolant are encouraged, as well as fuel property measurements to fill gaps in existing databases. Design, irradiation (both neutron and ion beam), or postirradiation examination of fuel concepts is beyond the scope of the call, but new methods of analyzing archival postirradiation data may constitute a component of proposed work. While the potential topic areas for the present call are broad, it will be critical that proposed work clearly articulate the key unknowns or challenges that limit further advancement of the chosen concept and how the proposed work will surmount those challenges if successful.

FC-2.3: HIGH-THROUGHPUT AND/OR MICRO-SCALE POST-IRRADIATION EXAMINATION TECHNIQUES TO SUPPORT ACCELERATED FUEL TESTING
(FEDERAL POC – KEN KELLAR & TECHNICAL POC – LUCA CAPRIOTTI)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $800,000)

There are current activities in the Advanced Fuels Campaign to perform irradiation testing on a sizable number (i.e., dozens) of miniature, significantly reduced-diameter fuel rodlets (e.g., metallic or oxide fuels with steel cladding diameters as low as 0.1-in. and cladding lengths on order of 2.0-in.) as part of an approach to significantly accelerate the burnup accumulation rate in the Advanced Test Reactor and simultaneously explore a diverse set of fuel design parameters. This approach will create the need to perform postirradiation examinations (PIE) on large numbers of miniature fuel rodlets, which will challenge traditional PIE methods relative to both timely throughput of large sample numbers and reduced dimensional scales. The present call intends to stimulate proposals to develop and demonstrate (on unirradiated, surrogate specimens) innovative techniques that can be applied in a hot cell environment to efficiently perform PIE on miniaturized integral fuel rods. Both non-destructive and destructive methods are of interest, including (but not limited to) measurements of cladding deformation, fuel swelling, fission gas release, and microstructural characterization of fuel or cladding.

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FC-2.4: MAINTAINING AND BUILDING UPON THE HALDEN LEGACY OF *IN-SITU* DIAGNOSTICS (FEDERAL POC – KEN KELLAR & TECHNICAL POC – COLBY JENSEN) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $800,000)

With the loss of access to the Halden reactor, the nuclear research community is at risk of losing the extensive *in situ* diagnostic capabilities practiced at Halden. The global nuclear community is involved in using and expanding upon Halden's diagnostic achievements. The Department of Energy, Office of Nuclear Energy is interested in research and development efforts that utilize and improve upon Halden in situ diagnostics. This call is open to all viable diagnostic approaches and is not limited to the methods used at the Halden facility. Halden also demonstrated excellence in test specimen manufacture. Their work included conversion of large irradiation specimens to smaller instrumented specimens while preserving important features of the test specimen, e.g., the cracked state of irradiated fuel pellets. Another notable attribute of Halden experiments was the ability to combine multiple diagnostics on one test specimen.

Real-time in-core diagnostic instrumentation of interest include, but are not limited to: creep, crack propagation, swelling, corrosion/crud build up, temperature, pressure, flux, two-flow phase, and fission product transport. Research that enables in-core application and associated logistics is also encouraged such as focuses on miniaturization, non-contact/non-intrusive as well as innovative data transmission techniques, such as wireless methods is also encouraged.

Emphasis in awarding R&D grants will be placed on diagnostics that can most directly benefit ongoing modelling and computer simulation development and future U.S. irradiation experiments, and that measure phenomena that is difficult to assess during irradiation or post-irradiation examinations, e.g., crack propagation rates and non-linear phenomena.

FC-2.5: SEPARATE EFFECTS TESTING IN TREAT USING STANDARD TEST CAPSULES (FEDERAL POC: KEN KELLAR & TECHNICAL POC: DAN WACHS) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (REFER TO NSUF WORKSCOPE TIME PERIODS (PART II, SECTION E.2.1), UP TO $500,000) (NSUF LETTER OF INTENT REQUIRED)

Modern nuclear fuel technology development strategies are being developed that rely on integration of micro-scale material science and thermal-mechanical engineering through advanced modeling and simulation techniques that are envisioned to accelerate the development and deployment of advanced fuel technologies. Realization of this vision requires implementing current and new nuclear science research tools in new ways to build the experimental databases that will be used to develop and qualify these tools. This includes developing the physical models input into the codes as well as the integral system data to be used in validating the result of the simulations.

TREAT is designed to deliver a time dependent neutron flux to test specimens. The specific shape of the transient (ranging through steady-state 'flat-tops', power ramps, pulses, or combinations of each) can be selected to achieve the experimenter's desired energy deposition in the test sample. The test can be immersed in a variety of sample environments provided by specialized irradiation devices. Multi-purpose, modular test devices allow experimenters the flexibility to quickly design experiment-specific test capsules that can provide a wide array of thermal, mechanical, and/or chemical environmental boundary conditions. The response of the test sample to the nuclear stimulus while immersed in this carefully controlled environment can be readily monitored using existing qualified or user supplied instruments.

Proposals are encouraged that will leverage TREAT's Minimal Activation Reusable Capsule Holder (MARCH) irradiation testing system and modern modeling and simulation tools to conduct novel separate effects tests of this type. Examples may include in-situ evaluation of physical properties of fissile material while under irradiation, thermal-mechanical response of fuel system components to nuclear heating, or short-term

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microstructural evolution of fissile materials under irradiation. Test samples can be supplied by the experimenter or allocated from the NSUF or DOE program's library of historic materials (fresh or pre-irradiated). Applicant should review the 2019 TREAT/NSUF awarded work to ensure new proposals are unique and/or complementary to the ongoing work.

NOTE: Access to NSUF capabilities will require agreement and final signature to the User Agreement (copy provided in Appendix D and at https://atrnsuf.inl.gov/documents/ATRNSUFStandardNon-PropUserAgreement.pdf). **The terms and conditions of the User Agreement are non-negotiable and failure to accept the terms and conditions of the User Agreement will terminate processing and review of the FC-2.5, NSUF-1, or NSUF-2 applications.** In order to ensure compliance throughout the application review process, applicants must indicate during the pre-application and full application submission that the User Agreement has been read, understood, and the terms and conditions are accepted. Further, submission of a pre-application and a full application indicates the applicant will comply and agree to the terms and conditions of the User Agreement. Upon award of an NSUF supported project, the User Agreement must be signed before activities will begin on the project.

MATERIALS PROTECTION, ACCOUNTING AND CONTROL TECHNOLOGY (FC-3)
(FEDERAL POC – MIKE REIM & TECHNICAL POC – MIKE BROWNE)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 2 YEARS AND $400,000)

The Materials Protection, Accounting and Control Technology program seeks to develop innovative technologies, analysis tools, and advanced integration methods to enable U.S. domestic nuclear materials control and accounting for advanced fuel cycles and reactors. This program also includes assessing vulnerabilities in current nuclear systems while minimizing proliferation and terrorism risks. Proposals are requested to develop innovative materials control and accounting technologies and tools for molten salt related nuclear energy applications. Technologies and tools should be able to increase the accuracy, reliability, and efficiency of nuclear materials quantification and/or tracking capability.

USED NUCLEAR FUEL DISPOSITION (FC-4)

FC-4.1: USED NUCLEAR FUEL DISPOSITION: DISPOSAL
(FEDERAL POC – JOHN ORCHARD & TECHNICAL POC – PETER SWIFT)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Assessments of nuclear waste disposal options start with waste package failure and waste form degradation and consequent mobilization of radionuclides, reactive transport through the near field environment (waste package and engineered barriers), and transport into and through the geosphere. Science, engineering, and technology improvements may advance our understanding of waste isolation in generic deep geologic environments and will facilitate the characterization of the natural system and the design of an effective engineered barrier system for a demonstrable safe total system performance of a disposal system. DOE is required to provide reasonable assurance that the disposal system isolates the waste over long timescales, such that engineered and natural systems work together to prevent or delay migration of waste components to the accessible environment.

Mined geologic repository projects and ongoing generic disposal system investigations generate business and R&D opportunities that focus on current technologies. DOE invites proposals involving novel material development, testing methods, and modeling concept and capability enhancements that support the program efforts to design, develop, and characterize the barrier systems and performance (i.e., to assess the safety of a nuclear waste repository). DOE will also consider proposals addressing applications of state-of-the-art uncertainty quantification and sensitivity analysis approaches to coupled-process modeling and performance assessment which contribute to a better assurance of barrier system performance and the optimization of

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

repository performance.

Research proposals are sought to support the development of materials, modeling tools, and data relevant to permanent disposal of spent nuclear fuel and high-level radioactive waste for a variety of generic mined disposal concepts in clay/shale, salt, crystalline rock, and tuff. Key university research contributions for the disposal portion of this activity may include one or more of the following:

- Improved understanding of waste package failure modes and material degradation processes (i.e. corrosion) for heat generating waste containers/packages considering direct interactions with canister and buffer materials in a repository environment leading to the development of improved models (including uncertainties) to represent the waste container/package long term performance.

- New concepts or approaches for alleviating potential post-closure criticality concerns related to the disposal of high capacity waste packages. Development of models and experimental approaches for including burn-up credit in the assessment of the potential for criticality assessment for spent nuclear fuel permanently disposed in dual- purpose canisters that are designed and licensed for storage and transportation only.

- Development of pertinent data and relevant understanding of aqueous speciation, multiphase barrier interactions, and surface sorption at elevated temperatures and geochemical conditions (e.g., high ionic strength) relevant to deep geologic disposal environments.

- Identification and assessment of innovative and novel buffer materials, new methods and tools for multi-scale integration of flow and transport data, new approaches for characterization of low permeability materials, state-of-the-art tools and methods for passive characterization and monitoring of engineered/natural system component properties and failure modes and their capability to isolate and contain waste.

FC-4.2: USED NUCLEAR FUEL DISPOSITION: STORAGE & TRANSPORTATION
(FEDERAL POC – JOHN ORCHARD & TECHNICAL POC – PETER SWIFT)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The possibility of stress corrosion cracking (SCC) in welded stainless steel dry storage canisters (DSC) for spent nuclear fuel (SNF) has been identified and studied as a potential safety concern. The welding procedure introduces high tensile residual stress and microstructure sensitization in the heat-affected zone (HAZ) of the fabrication welds in the DSC, which might promote the initiation of pitting and transition to SCC growth when exposed to an aggressive chemical environment. Further, if storing the canister in a marine environment, the atmospheric salts might deliquesce and generate an aqueous brine layer on the surface of the canisters at various locations, creating an aggressive chemical environment that could lead to pitting and subsequent SCC. Studies and research conducted so far indicate that the possibility of SCC in these canisters is low and that implementing technologies to mitigate SCC might further reduce the risks for long-term storage of these DSCs.

Research proposals are sought to develop mitigation technologies for enhancing the reliability of long-term storage and maintenance of DSCs. Possible technologies include sprays, coatings, weld depositions, and other techniques where long-term performance can be demonstrated.

MISSION SUPPORTING: FUEL CYCLE TECHNOLOGIES
MS-FC-1: UNDERSTANDING, PREDICTING, AND OPTIMIZING THE PHYSICAL PROPERTIES, STRUCTURE, AND DYNAMICS OF MOLTEN SALT **(FEDERAL POC – STEPHEN KUNG & TECHNICAL POC – MARK WILLIAMSON)** **(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)** **(UP TO 2 YEARS AND $400,000)** Molten salts find applications in advanced nuclear technologies as electrolytes for pyro-processing and fuel solvents and coolants for advanced reactors. Thermodynamic models are needed to predict critical salt characteristics such as melting points, heat capacity, free energies for potential corrosion reactions, and solubilities for fission and corrosion products as function of temperature and composition. The atomic composition and redox condition of the salt may change with of time as a result of fission product formation and irradiation effect. Proposals are requested to better understand, predict, and optimize the physical properties and thermochemical behavior of molten salts. The goal is to develop and use first-principles molecular dynamics simulations and computational electronic structure method to extend the limited experimental data sets in covering a broader range of chemical evolution and environments. Innovative approaches to (1) apply molecular dynamics simulations to predict thermophysical and transport properties; (2) build multi-component models for prediction of phase diagrams; and (3) develop advanced models to guide the experimental efforts to manipulate the molten salt thermophysical properties are especially encouraged. **MS-FC-2: UNDERSTANDING THE STRUCTURE AND SPECIATION OF MOLTEN SALT AT THE ATOMIC AND MOLECULAR SCALE** **(FEDERAL POC – STEPHEN KUNG & TECHNICAL POC – MARK WILLIAMSON)** **(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)** **(UP TO 3 YEARS AND $600,000)** To understand the effects of structure and dynamics of molten salts on their physical and chemical properties—such as viscosity, solubility, volatility, and thermal conductivity—it is necessary to determine the speciation of salt components as well as the local and intermediate structure at operationally relevant temperatures. Real-time spectroscopic and electrochemical methods can help monitoring key chemical species in solution. Proposals are requested to take advantage of recent breakthroughs in advanced characterization tools and instrumentation methods to provide information at the atomic and molecular scale. The goals are to determine the local structure and bonding of chemical species in salt solution and to develop innovative real-time analytical methods for microscopic and macroscopic property measurements. Innovative approaches to: (1) determine salt molecular structure using scattering and spectroscopic methods; (2) develop novel electrochemistry and spectroscopy methods for in-situ monitoring and predictive modeling; and (3) develop molten salts optical basicity scale to determine corrosivity and solubility of actinides are especially encouraged.

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION (NEAMS)

NEAMS-1: DEVELOPMENT OF GENERALIZED MULTIGROUP CROSS SECTIONS FOR ARBITRARY REACTOR GEOMETRIES

(FEDERAL POC: DAVE HENDERSON & TECHNICAL POC: DAVID KROPACZEK)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $750,000)

The generation of multigroup cross sections for light water reactor (LWR) geometry is a fairly well understood problem. The solution generally relies on well-known geometry factors for circular fuel rods arranged in square lattices. However, many advanced reactor designs being proposed no longer use circular fuel rods in square lattices. New reactor designs include: TRISO double heterogeneous fuel, Light Bridge fuel with "clover" designs, TCR fuel with square fuel segments with round flow channels, and molten salt reactors with homogeneous fuel. In addition to the geometry issues, many advanced designs have energy spectrums in epi-thermal and fast ranges. The main difficulty in calculating accurate multigroup cross sections is calculating appropriate energy spectrums of resonances in the arbitrary geometry. The current approach of handling arbitrary geometry configurations and different energy spectrums is to use Monte Carlo (MC) methods with continuous energy cross sections. While this approach is general in nature, MC methods are not efficient for design work involving either large reactors and/or 1000's of different design calculations. The inefficiency of MC methods is amplified by multi-physics feedback, where the temperatures and material densities are additional parameters to solve for.

Proposals are sought that investigate and develop efficient methods of generating multigroup cross sections for reactors with arbitrary geometries and energy spectrums. The primary focus should be the generation of cross sections in the resolved resonance ranges. The proposed methods must also be able to support multi-physics feedback. The ability to calculate multigroup cross sections for arbitrary geometries and energies will greatly enhance the ability to perform efficient core design calculations.

The proposal should include the development of multiple benchmark problems covering both existing LWR geometries and proposed advanced reactor configurations. The final results should include side-by-side benchmark results run with MC methods, current methods (where applicable), and any new methods. Results should focus on both accuracy and computer run-times

NEAMS-2: NEAR-WALL GAS-FLOW CORRELATIONS IN PEBBLE BED REACTORS

(FEDERAL POC: DAVE HENDERSON & TECHNICAL POC: RICHARD MARTINEAU)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $750,000)

The pebble bed reactor (PBR) is a type of gas-cooled reactor that combines a variety of desirable features such as inherent meltdown safety, small excess reactivity, and online refueling. Design and licensing of pebble bed reactors almost exclusively relies on the porous medium flow approximation. The accuracy of porous media calculations hinges on the accuracy of the correlations for pressure drop, effective conductivity of the pebble bed, and heat transfer coefficients from the pebbles to the coolant.

One of the phenomena of interest is flow near the wall of the pebble bed and its interaction with core bypass flow through gaps between reflector blocks. Currently existing correlations are insufficient for an accurate prediction of these flow conditions because measured data is extracted from integral pressure drop experiments, for isothermal flow conditions, and/or neglecting the effect of flow into/out of reflector gaps. Despite an "high importance" ranking in the phenomena identification and ranking tables, near-wall flow and its interaction with bypass flow remains poorly understood.

Improving the understanding the important effects for flow near the wall of a porous medium and its interaction with core bypass flow is of great importance for PBRs. The desired primary outcome from these investigations is

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION (NEAMS)

the generation of new and useful correlations for pressure drop and heat transfer coefficients between pebbles, wall, and fluid. Correlations must be useable in porous media codes such as Pronghorn.

We seek an experimental investigation of flow near the pebble wall interface based on high-resolution, spatio-temporal resolved measurements techniques such as time-resolved particle image velocimetry (TR-PIV), laser induced fluorescence (LIF), and matched refractive index (MRI). Experimental techniques should be paired with advanced data analysis techniques such as proper orthogonal decomposition, image recognition, or other suitable methods that allow extraction of pertinent information from the flow field. It is desired to provide for non-uniform pebble heating and adjustable core bypass flow in the experimental apparatus and test matrix. In support of the experimental work, high fidelity calculation with advanced LES and URANS CFD software such as Nek5000 is recommended for supporting conclusions, computing non-measurable quantities, and extrapolating the obtained data.

NEAMS-3: MOLTEN SALT CHEMISTRY MODELING
(FEDERAL POC: DAVE HENDERSON & TECHNICAL POC: DAVID ANDERSSON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $750,000)

Design and deployment of molten salt reactors require understanding of salt chemistry and corrosion of structural materials used in reactor components, where corrosion is interpreted broadly to include both loss of material from the structural component and deposition of chemical species in the salt. The salt chemistry combined with the choice of alloy for structural materials determines the corrosion potential, which, together with kinetic coefficients, governs the rate of material loss or gain. In addition to the chemistry of a freshly fueled pure salt, it is also critical to determine the impact of impurities, additives and fission products, since they often have an outsized influence on the corrosion potential and thus the real-life material performance. The chemical state of each species in the salt is also important to assess the consequences of a potential release of fuel or fission products in an accident scenario. Modeling these behaviors requires thermodynamic databases of fuel salts with dissolved impurities, additives and fission products as well as for the relevant structural alloys. The biggest capability gaps exist for the salt chemistry. Some of the information needed to expand the salt databases can be found in the literature and new data is being acquired or is planned to be acquired through experiments and atomic scale simulations based on either density functional theory or empirical potentials. In order to enable predictions of molten salt chemistry and corrosion, work is solicited to synthesize new thermodynamic and corrosion data into high-quality validated databases that can be used for complex multi-component salts and that include uncertainty quantification. Support for filling identified data gaps and reducing uncertainties with targeted experiments and/or modeling in coordination with national laboratories and other researchers funded separately through existing DOE NE projects may also be considered. Work on interfacing the thermodynamic databases with corrosion models specifically incorporating the micro-structure of alloys used in structural components is also encouraged. Proposals must clearly identify the specific salt to be used, neutron spectrum, facilities to be used, schedule and the portion of the above scope to be conducted.

MISSION SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

MS-NEAMS-1: SAM BASELINE DEVELOPMENT
(FEDERAL POC – DAVE HENDERSON & TECHNICAL POC – TANJU SOFU)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 18 MONTHS AND $300,000)

1. Multi-scale thermal fluid simulation

Multi-scale thermal fluid simulations are crucial for a wide range of transient simulations of nuclear systems. New algorithm or schemes for coupling at 1D-3D fluid-to-fluid interfaces or 2D-3D structure-to-fluid interfaces are sought for. New experimentation designed explicitly for validations of multi-scale thermal fluid simulation, analysis of existing benchmark datasets or development of new benchmark datasets, as well as performing direct comparison of datasets with NEAMS code simulations are also of interests.

2. Integrate the material transport model into SAM

Material performance under high temperature irradiative and corrosive environments remains a key challenge for advanced reactor applications. It is important to integrate computationally efficient yet accurate lumped parameter material models into system-level analysis code SAM for transient safety evaluations of advanced reactors such as liquid-metal- or salt-cooled reactors. Developing advanced reduced-order material models and integrating them (including the phase changes, source term, precipitation and corrosion) into the system-level code such as SAM are of interests.

PROGRAM SUPPORTING: NUCLEAR ENERGY

NUCLEAR ENERGY-CYBERSECURITY RESEARCH TOPICS AND METRICS ANALYSES (NE-1)
(FEDERAL POC: SUIBEL SCHUPPNER & TECHNICAL POC: STEVEN HARTENSTEIN)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Cost-effectively preventing, detecting, and mitigating cyber threats to nuclear energy systems is the subject of this research. Understanding the risks associated with each design decision is fundamental to cyber protection. With the increasing application of digital instrumentation, control, and communication systems and the constant evolution of cyber security threats and technologies, there is a need for comprehensive analytical capability to model and simulate control systems and their vulnerabilities.

Proposals are sought for modeling and simulation capabilities that can inform researchers, designers and operators when assessing cyber security risks. Research of most interest will addresses characteristics and behaviors of components within embedded instrumentation and control (I&C) systems that are used within the nuclear enterprise.

Another area of interest includes Integration of cyber research enabling platforms that would couple high fidelity nuclear plant simulators with emulation, hardware-in-the-loop, or human-in-the-loop instances of control and communication. Models shall capture the behavior of an I&C system, to 1) simulate characteristics of an I&C system under cyber-attack; 2) study the cyber risk impacts of upgrades and maintenance on such systems; 3) enable future nuclear energy cyber security research, and 4) facilitate nuclear facility operation education and training.

INTEGRATED ENERGY SYSTEMS DESIGN AND MODELING (NE-2)
(FEDERAL POC: MELISSA BATES & TECHNICAL POC: SHANNON BRAGG-SITTON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Advanced nuclear-renewable integrated energy systems (IES) are comprised of one or more nuclear and renewable energy sources, industrial energy users, and energy storage systems. Various IES configurations are being evaluated for their economic benefit and technical feasibility within various geographic regions; systems may be "tightly coupled" within a single "energy park" type of configuration, or may be loosely coupled within a grid balancing area.

Work scopes of interest for FY20 proposals focus on the use of IES to advance transformational technology and innovation to meet the global need for safe, secure, and affordable water. The Water Security Grand Challenge is of particular interest for application of IES (see https://www.energy.gov/eere/water-security-grand-challenge). The Water Security Grand Challenge is a White House initiated, U.S. Department of Energy led framework to advance transformational technology and innovation to meet the global need for safe, secure, and affordable water. Key goals within the challenge include:

- "Launch desalination technologies that deliver cost-competitive clean water,"
- "Transform the energy sector's produced water from a waste to a resource,"
- "Achieve near-zero water impact for new thermoelectric power plants, and significantly lower freshwater use intensity within the existing fleet," and
- "Develop small, modular energy-water systems for urban, rural, tribal, national security, and disaster response settings."

For example, the petroleum extraction produces a large volume of waste water. Processes for the purification of this waste water, and the coupling of such processes to nuclear thermal energy sources (current fleet, light-water small modular reactors, or other advanced reactor designs) are of interest. Researchers should not propose a novel reactor design, but should instead focus on novel coupling technologies, system control, and optimized operational dispatch to support both water processing and electricity production demands. Cyber-informed engineering should be considered in system design, dispatch optimization, and system control. Computational models should be capable of integrating with the Modelica and RAVEN-based ecosystem for modeling and analysis that is used by the lab team for IES. Systems of interest could be applicable to fixed installations or could be modular and transportable in their design to address the range of applications called out within the Water Security Challenge. Principle investigators are encouraged to investigate the potential markets and market competitiveness of proposed solutions within large-scale centralized grid or islanded microgrids that may be applicable to SMR or microreactor technologies.

TRANFORMATIONAL CHALLENGE REACTOR R&D (NE-3)
(FEDERAL POC: TANSEL SELEKLER & TECHNICAL POC: TBD)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

MISSION SUPPORTING: NUCLEAR ENERGY

INTEGRAL BENCHMARK EVALUATIONS (MS-NE-1)
(FEDERAL POC: DAVE HENDERSON & TECHNICAL POC: JOHN BESS)
(UP TO 3 YEARS AND $400,000)

The International Reactor Physics Experiment Evaluation Project (IRPhEP) and International Criticality Safety Benchmark Evaluation Project (ICSBEP) are recognized world-class programs that have provided quality assured (peer-reviewed) integral benchmark specifications for thousands of experiments. The Project produces two annually updated Organization for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) Handbooks that are among the most frequently quoted references in the nuclear industry. Applications are sought, within the scope of these two projects, to provide complete benchmark evaluations of existing experimental data that would be included in IRPhEP and ICSBEP handbooks, and would support current and future R&D activities.

The IRPhEP and ICSBEP Handbooks are the collaborative efforts of nearly 500 scientists from 24 countries to compile new and legacy experimental data generated worldwide. Without careful data evaluation, peer review, and formal documentation, legacy data are in jeopardy of being lost and reproducing those experiments would incur an enormous and unnecessary cost. The handbooks are used worldwide by specialists in reactor safety and design, criticality safety, nuclear data, and analytical methods development to perform necessary validations of computational models. Proposed benchmark evaluations should be of existing experimental data. Measurements of interest include critical, subcritical, buckling, spectral characteristics, reactivity effects, reactivity coefficients, kinetics, reaction-rate and power distributions, and other miscellaneous types of neutron and gamma transport measurements. A growing area of interest includes evaluation of transient and/or multiphysics benchmark experiment data for light water reactor systems, such as PWRs and BWRs.

All evaluations must be completed according to the requirements, including peer review, in the IRPhEP and the ICSBEP. DOE currently invests tens of millions of dollars each year to develop the next generation of nuclear engineering modeling & simulation tools. These tools need ad-hoc evaluated and quality-assured experimental data for validation purposes and, consequently, benchmark evaluations in support of DOE programs such as, but not limited to, TREAT, LWRS, FCT, ART, and NE's Advanced Modeling and Simulation Program (which combines application of computational capabilities from the NEAMS ToolKit and the VERA suite developed by the Energy Innovation Hub for Reactor M&S) are of particular interest to this call.

NUCLEAR DATA NEEDS FOR NUCLEAR ENERGY APPLICATIONS (MS-NE-2)
(FEDERAL POC: DAVE HENDERSON & TECHNICAL POC: TBD)
(UP TO 3 YEARS AND $400,000)

The Evaluated Nuclear Data File (ENDF) maintained by the National Nuclear Data Program (NNDC) at Brookhaven National Laboratory (BNL) provides the most reliable and commonly used nuclear data for nuclear energy applications. However, a close and critical examination of the existing nuclear data often finds that it is inadequate for current and emerging applications.

Proposals are sought that address nuclear data needs in NE mission areas, provided that these needs are clearly demonstrated to be a limiting factor in nuclear fuel and reactor design, analysis, safety, and licensing calculations. Use of sensitivity and uncertainty analysis methods in proposed efforts is encouraged to demonstrate these needs.

Many nuclear data needs for NE may be found in the NEA Nuclear Data High Priority Request List (HPRL) (https://www.oecd-nea.org/dbdata/hprl/, which includes a broad spectrum of needs encompassing light water reactors (LWRs) as well as sodium fast reactors. Other emerging needs not yet listed on the HPRL include continued investigations of thermal scattering data in high-temperature graphite, thermal scattering data for fluorine-based molten salt reactors, and chlorine reactions for fast spectrum molten salt reactors. Additional nuclear data needs that meet documented needs for industry and DOE-NE missions are also encouraged especially as aligned with the Gateway for Accelerated Innovation in Nuclear (GAIN), Nuclear Energy Advanced Modeling and Simulation (NEAMS), Consortium for Advanced Simulation of LWRs (CASL), Advanced Reactor Technologies (ART), Fuel Cycle Research and Development (FCR&D), Transient Test Reactor (TREAT), Light Water Reactor Sustainability (LWRS) and others.

Proposals are sought that provide relevant improvements in nuclear data that address one or more stated needs by developing and demonstrating the enhancements through the entire nuclear data pipeline, from 1) new nuclear data measurements; 2) evaluation in the appropriate format (e.g. ENDF); 3) inclusion of nuclear data covariances; 4) processing into usable forms for application codes; 5) confirmation of improved predictions and uncertainties through application studies and validation; and 6) deployment through the National Nuclear Data Center at BNL for inclusion by external users in quality-assured design, analysis, safety, and licensing calculations. Partnerships with national laboratories and especially industry to clearly articulate the need for the data and to demonstrate the use of improved data in production applications are strongly encouraged.

MISSION SUPPORTING GRAND CHALLENGE (MS-NE-3)
(FEDERAL POC – Brad Williams & TECHNICAL POC – John Carmack) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $400,000)

The Office of Nuclear Energy mission supports enhancing the long-term viability of the presently operating light water reactors, stimulating the development and commercialization of advanced reactor concepts, and extending nuclear energy beyond conventional electrical generation applications. Many specific challenges have been identified elsewhere in this FOA, yet many challenges remain. Applications are sought that address other issues that hinder continued operation of the existing fleet, improve the deployment potential of advanced reactor concepts or expand nuclear energy's role in meeting the nation's energy, environmental, and national security needs. Applicants must clearly outline the challenge to be addressed, the proposed solution, and the methodology that will be used to achieve the solution, including specific resources (costs and schedules) and milestones associated with the proposal's activities, as well as estimates of longer-term resources (costs and schedules) and milestones associated with implementation of the proposed solution. Proposed solutions can be at the system or component level. High-risk, high-reward ideas are encouraged.

Appendix B: Workscopes for U.S. University-, National Laboratory-, or Industry-led* Program Supporting R&D Projects

*Industry may only lead in NSUF workscopes

PROGRAM SUPPORTING: NUCLEAR ENERGY ENABLING TECHNOLOGIES (NEET)

ADVANCED METHODS FOR MANUFACTURING (NEET-1)
(FEDERAL POC – TANSEL SELEKLER & TECHNICAL POC – BRUCE LANDREY)
(ELIGIBLE TO LEAD: UNIVERSITY OR NATIONAL LABORATORY)
(UP TO 3 YEARS AND $1,000,000)

The Advanced Methods for Manufacturing program seeks proposals for research and technology development to improve the methods by which nuclear equipment, components, and plants are manufactured, fabricated, and assembled. Proposals should support the Department of Energy's (DOE) Office of Nuclear Energy's (NE) mission to advance U.S. nuclear power in order to meet the nation's energy needs by: 1) enhancing the long-term viability and competitiveness of the existing U.S. reactor fleet; 2) developing an advanced reactor pipeline, and, 3) implementing and maintaining the national strategic fuel cycle and supply chain infrastructure.

The goal of the program is to accelerate innovations that reduce the cost and schedule of constructing new nuclear plants and make fabrication of nuclear power plant components faster, cheaper, and more reliable. The program seeks to encourage innovation that supports the "factory fabrication" and expeditious deployment of reactor technologies. Potential areas for exploration include:

1.1 FACTORY AND FIELD FABRICATION TECHNIQUES

Applications are sought for innovative technologies specific to improvements in surface modification and cladding techniques, and advanced modular factory and field fabrication and installation techniques.

1.2 QUALITY CONTROL TECHNIQUES AND QUALIFICATION METHODOLOGIES

Applications are sought to develop quality control techniques and qualification methodologies for advanced manufacturing processes. This should include active engagement with consensus standard organizations with a pathway to code/regulatory acceptance.

The most up-to-date information on active AMM projects can be found in the 2019 NEET Advanced Methods for Manufacturing Awards Summaries on the NE website under NEET documents.

ADVANCED SENSORS AND INSTRUMENTATION (NEET-2)

The Advanced Sensors and Instrumentation program seeks applications for innovative technology for controls, analytics, and instrumentation of advanced reactors systems. Technology should demonstrate greater accuracy, reliability, resilience, higher resolution, and ease of replacement/upgrade capability for applications in the nuclear environment, minimizes operations and maintenance costs, and address regulatory concerns.

The proposal should indicate whether and how the proposed technology is or may be applicable to multiple reactors or fuel cycle applications, i.e. crosscutting. Proposals should support the Department of Energy's (DOE) Office of Nuclear Energy's (NE) mission to advance U.S. nuclear power in order to meet the nation's energy needs by: 1) enhancing the long-term viability and competitiveness of the existing U.S. reactor fleet; 2) developing an advanced reactor pipeline, and, 3) implementing and maintaining the national strategic fuel cycle and supply chain infrastructure.

NEET-2.1: ADVANCED CONTROL SYSTEMS
(FEDERAL POC – SUIBEL SCHUPPNER & TECHNICAL POC – CRAIG PRIMER)
(ELIGIBLE TO LEAD: UNIVERSITY OR NATIONAL LABORATORY)
(UP TO 3 YEARS AND $1,000,000)

Applications are sought for research projects that will design, develop, and demonstrate advanced control for semi-autonomous and remote operation of advanced reactor designs. Outcomes should:

PROGRAM SUPPORTING: NUCLEAR ENERGY ENABLING TECHNOLOGIES (NEET)

- Streamline and simplify the advanced control system design process to automate and enhance plant operation. Design should result in reduced long term Operations and Maintenance costs and include a cost benefit analysis.
- Reduce I&C testing, validation and verification efforts associated with licensing requirements through developing verifiable architectures (hardware and software) using novel test methodologies that would enable use of smart digital devices in both safety related and non-safety related applications for nuclear power plants.

NEET-2.2: BIG DATA, MACHINE LEARNING, AND ARTIFICIAL INTELLIGENCE
(FEDERAL POC – SUIBEL SCHUPPNER & TECHNICAL POC – CRAIG PRIMER)
(ELIGIBLE TO LEAD: UNIVERSITY OR NATIONAL LABORATORY)
(UP TO 3 YEARS AND $1,000,000)

Applications are sought to develop and demonstrate advanced analytics for nuclear plant operation and maintenance systems that support semi-autonomous and remote monitoring of advanced reactor designs. Cost-benefit analysis should be conducted as part of the project to demonstrate technology or product viability. Research should:

- Demonstrate an optimal balance between cost and plant performance to achieve reliability, availability, maintainability, and security.

Develop and demonstrate a transformational approach to monitor and analyze semi-autonomous operation of advanced reactors through real time integration of predictive analytics and risk informed condition monitoring.

NEET-2.3: ADVANCED SENSORS AND COMMUNICATION
(FEDERAL POC – SUIBEL SCHUPPNER & TECHNICAL POC – CRAIG PRIMER)
(ELIGIBLE TO LEAD: UNIVERSITY OR NATIONAL LABORATORY)
(UP TO 3 YEARS AND $1,000,000)

Applications are sought to enable deployment of sensors, instrumentation, and supporting electronics for advanced reactor concepts, with a particular interest in technologies that would enable semi-autonomous and remote operation. Applicants should:

- Assess and demonstrate the impact of the selected technology toward remote and semi-autonomous operation of advanced reactor concepts.
- Develop radiation hardened electronic systems to support wired and wireless communication of data from sensors and instrumentation located in-vessel and near-vessel in high temperature and high radiation environments found in advanced reactors.
- Develop sensor performance models for the initial calibration, routine operation, radiation degradation, in-situ re-calibration, and predictive failure mechanisms to understand sensor performance for the lifetime of operation in advanced reactors.

PROGRAM SUPPORTING: NUCLEAR SCIENCE USER FACILITIES (NSUF-1)

NUCLEAR ENERGY-RELATED R&D SUPPORTED BY NUCLEAR SCIENCE USER FACILITIES CAPABILITIES (NSUF-1)

NOTE: FC-2.5 require NSUF access but can only be led by universities. Those workscopes can be found in Appendix A under the “Fuel Cycle” header.

This workscope solicits applications for nuclear energy-related research projects focused on the topical areas described below. It is intended that these focused topical areas will change with each future CINR FOA. The focused topical areas are selected by NE’s R&D programs (e.g. Nuclear Reactor Technologies, Fuel Cycle Technologies, and Nuclear Energy Enabling Technologies) with the explicit purpose to leverage the limited R&D funding available with access to NSUF capabilities. All applications submitted under this workscope will be projects coupling R&D funding with NSUF access. Projects requiring “NSUF access only” (see NSUF-2 below) or “R&D funding only” must be submitted under other appropriate workscopes. Applications submitted under this workscope must support the Department of Energy Office of Nuclear Energy mission. Capabilities available through the NSUF can be found on the website at nsuf.inl.gov.

The Office of Nuclear Energy (NE) supports the Department of Energy’s HPC4 Materials (High Performance Computing for Materials) initiative to accelerate "industry discovery, design, and development of materials for severe environments by enabling access to computational capabilities and expertise in the DOE laboratories". NE's high-performance computing capabilities include Falcon at the Idaho National Laboratory. More information on computational resources can be found at NSUF.inl.gov. NE is seeking proposals for the development of innovative materials or material concepts for the extreme operating and accident environments expected in advanced reactor and fuel cycle technologies using the high-performance computing capabilities at the INL.

NSUF 1.1: TESTING OF ADVANCED MATERIALS OR ADVANCED SENSORS FOR NUCLEAR APPLICATIONS
(FEDERAL POC: SUIBEL SCHUPPNER & TECHNICAL POC: BRENDEN HEIDRICH)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(UP TO 7 YEARS AND $500,000)

Proposals are sought for irradiation testing and post-irradiation examinations that support the development of advanced materials for sensors, and development of advanced sensors themselves to support NE’s mission to enhance the long term viability and competitiveness of the existing fleet, to develop an advanced reactor pipeline, and to implement and maintain national strategic fuel cycle and supply chain infrastructure. **This funding does not support research and development activities to develop materials or sensors, but rather the irradiation of sensors and materials as described below.**

1) Advanced Materials for Sensors: Successful irradiation testing and post irradiation examination of candidate materials proposed for advanced sensors applications will include: a description of the materials; irradiation and post irradiation examination needs; the role of the materials in new sensors, controls, communications or associated applications.

2) Advanced Sensors: Successful irradiation and post irradiation examination of sensors and associated instrumentation will include: a description of the sensor and associated instrumentation and materials requiring irradiation and post irradiation examination; irradiation and post irradiation examination needs; and the purpose and application of the developed sensor in nuclear energy systems.

PROGRAM SUPPORTING: NUCLEAR SCIENCE USER FACILITIES (NSUF-1)

NSUF 1.2: IRRADIATION TESTING OF MATERIALS PRODUCED BY INNOVATIVE MANUFACTURING TECHNIQUES

(FEDERAL POC: TANSEL SELEKLER & TECHNICAL POC: BRUCE LANDREY)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(REFER TO NSUF WORKSCOPE TIME PERIODS (PART II, SECTION E.2.1), UP TO $500,000)

Products from advanced and innovative manufacturing, welding / joining, and surface modification and cladding techniques can be proposed for evaluation of irradiation effects on material performance in support of NE's mission to enhance the long term viability and competitiveness of the existing fleet, to develop an advanced reactor pipeline, and to implement and maintain national strategic fuel cycle and supply chain infrastructure.

This funding does not support research and development activities to develop manufacturing and construction techniques, but rather evaluate the irradiation effects on material performance.

NSUF 1.3: NUCLEAR MATERIALS DISCOVERY AND QUALIFICATION INITIATIVE

(FEDERAL POC: TANSEL SELEKLER & TECHNICAL POC: RORY KENNEDY)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(REFER TO NSUF WORKSCOPE TIME PERIODS (PART II, SECTION E.2.1), UP TO $500,000)

The NSUF has been tasked to lead the Nuclear Materials Discovery and Qualification Initiative (NMDQi) whose goal is to accelerate both the discovery of new materials that can be applied to the needs of the nuclear industry and the ultimate qualification of those materials. In order to facilitate this acceleration, the NSUF intends to follow the materials design concept and apply the combinatorial and high-throughput methodology (CHT) to the unique challenges of the nuclear materials field. The CHT methodology integrates combinatorial materials fabrication methods, high-throughput characterization techniques, material modeling, and data analytics with the potential of incorporating machine learning and artificial intelligence schemes. As a reference, potential proposers are directed to the Materials Genome Initiative and its successful use of the CHT methodology to introduce new materials into the market faster and at lower cost. Proposals are sought in any of the areas of the CHT methodology as applied to nuclear materials from fabrication to irradiation and PIE. In addition, proposals focused on the development of techniques for CHT, particularly in the area of high-throughput mechanical property testing and the high-throughput testing of radioactive materials will also be accepted. To this, CHT fabrication and testing techniques for bulk material properties as well as studies to correlate bulk properties to those obtained from CHT techniques on micro- or nano-scale samples are encouraged. Areas of interest include materials for core, cladding, and structural applications, metallic and ceramic advanced fuels, sensor materials, multi-layer structures (e.g. coatings), interface interactions (fcci, fcmi), and corrosion. All proposals must be directly associated with one or more of the CHT methodology components (combinatorial materials fabrication methods, high-throughput characterization techniques, material modeling, and data analytics).
In addition to the access to NSUF capabilities (including high performance computing), the NSUF will support the R&D activities of awardees for this work scope only.

NUCLEAR SCIENCE USER FACILITIES (NSUF-2)

NUCLEAR SCIENCE USER FACILITIES ACCESS ONLY (NSUF-2)

(FEDERAL POC: TANSEL SELEKLER & TECHNICAL POC: RORY KENNEDY)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, AND INDUSTRY)
(REFER TO NSUF WORKSCOPE TIME PERIODS (PART II, SECTION E.2.1))

Applicants interested in utilizing Nuclear Science User Facilities (NSUF) capabilities only should submit "access only" applications under this workscope. Applications must support the Department of Energy Office of Nuclear Energy's mission. Capabilities available through the NSUF can be found on the website at nsuf.inl.gov.

The Office of Nuclear Energy (NE) supports the Department of Energy's HPC4 Materials (High Performance Computing for Materials) initiative to accelerate "industry discovery, design, and development of materials for severe environments by enabling access to computational capabilities and expertise in the DOE laboratories". NE's high-performance computing capabilities include Falcon and Lemhi at Idaho National Laboratory. More information on computational resources available through NSUF can be found at https://nsuf.inl.gov/. NE is seeking proposals for the development of innovative materials or material concepts for the extreme operating environments expected in advanced reactor and fuel cycle technologies using the high-performance computing capabilities at the INL.
Experiments with synchrotron radiation may be proposed in applicable workscopes below. NSUF has access to the 6% of the available beam time at the X-ray Powder Diffraction beamline at NSLS-II.

NSUF-2.1: Core and Structural Materials

This element is primarily focused on fundamental understanding of irradiation effects in core and structural materials such as material aging and degradation mechanisms (e.g. fatigue, embrittlement, void swelling, fracture toughness, IASCC processes and mitigation), as well as developing alternate and/or radiation resistant materials for application in current and future fission reactors, and materials from alternate or advanced manufacturing techniques (including welding and joining). Proposed projects may involve R&D in the areas of material irradiation performance and combined effects of irradiation and environment on materials. Projects whose relevancy is based solely or primarily on fusion energy needs will not be considered. Proposals coupling experimental methods with modeling and simulation are highly encouraged.

NSUF-2.2: Nuclear Fuel Behavior and Advanced Nuclear Fuel Development

This program element is primarily focused on increasing our fundamental understanding of the behavior of nuclear fuels (including cladding) in reactor and research and development activities for advanced nuclear fuels and improving the performance of current fuels. Areas of interest include physics and chemistry of nuclear fuels, irradiation and thermal effects on microstructure development and the effects on, for example, thermophysical and thermomechanical properties as well as chemical interactions. Advanced fuels applicability extends to fast spectrum transmutation systems, coated particle fuels for high-temperature reactor systems, and robust fuels for light water reactors including accident tolerant fuels. Activities should be aimed at irradiation experiments and post irradiation examination that investigate fundamental aspects of fuel performance such as radiation damage, amorphization, fuel restructuring, species diffusion and migration, and fission product behavior. Separate effects testing focused on specific V&V issues are encouraged. Proposals coupling experimental methods with modeling and simulation are highly encouraged.

NSUF-2.3: Advanced In-reactor Instrumentation

This program element includes irradiation to support qualification of advanced in-reactor instrumentation for characterization of materials under irradiation in test reactors and for on-line core monitoring in power reactors. Applications should address the deployment and qualification strategy of radiation resistant sensors.

Development of techniques that are non-intrusive with respect to irradiation specimens, and nontraditional methods such as optical fibers and ultrasonic techniques as well as other incorporated wireless transmission techniques are encouraged. Proposals that also support the GAIN initiative, such as those involving development of advanced instrumentation, sensors, and measurement techniques for use in advanced reactors including molten salt reactors, sodium cooled fast reactors, lead cooled fast reactors, or high temperature gas reactors are encouraged. For MSR with dissolved fuel, an important and challenging problem is the ability to measure local chemical composition in real time at critical locations.

NSUF-2.4: HIGH PERFORMANCE COMPUTING AT IDAHO NATIONAL LABORATORY (LIMITED TO 3 YEARS)

Nuclear Science User Facility (NSUF) High-Performance Computing (HPC) resources offered through **Idaho National Laboratory** provide scientific computing capabilities to support efforts in advanced modeling and simulation. Proposals in this area may address a wide range of research activities, including performance of materials in harsh environments (including the effects of irradiation and high temperatures), performance of existing light water and advanced nuclear reactors, and multiscale multiphysics analysis of nuclear fuel performance.

Current HPC resources include:

- Falcon: a SGI ICE-X distributed memory system with 34,992 cores, 121 TB of memory and a LINPACK rating of 1 Petaflop/s. Falcon's network is a seven-dimensional enhanced hypercube utilizing FDR InfiniBand. Individual compute nodes contain dual Xeon E5-2695 v4 processors with 18 cores each and 128GB of memory. Falcon came online in Fall 2014 and ranked #97 on the November 2014 TOP500 list.
- Lehmi: a Dell 6420-based system with 20,160 cores, 94 TB of memory and a LINPACK rating of 1 Petaflop/s. Lemhi's network is an omnipath fat tree. Individual compute nodes contain dual Xeon Gold 6148 processors with 20 cores each and 192GB of memory. Lemhi came online in Fall 2018 and ranked #427 on the November 2018 TOP500 list.

HPC support includes: access to INL HPC systems, assistance with system login and running code, basic HPC training, and software support and expertise as requested. Software includes an assortment of tools in the areas of: Compter Aided Engineering, Chemistry, Code Development, Data Manipulation, Math, MOOSE, MPI, Neutronics and Transport, Numerical Libraries, Programming, and Visualization. Access to HPC resources through this FOA does not provide licenses to software. Use of DOE-developed software from the NEAMS and CASL programs is encouraged.

NOTE: Access to NSUF capabilities will require agreement and final signature to the User Agreement (copy provided in Appendix D. The terms and conditions of the User Agreement are non-negotiable and failure to accept the terms and conditions of the User Agreement will terminate processing and review of the FC-2.5, NSUF-1, or NSUF-2 applications. In order to ensure compliance throughout the application review process, applicants must indicate during the pre-application and full application submission that the User Agreement has been read, understood, and the terms and conditions are accepted. Further, submission of an pre-application and a full application indicates the applicant will comply and agree to the terms and conditions of the User Agreement. Upon award of an NSUF supported project, the User Agreement must be signed before activities will begin on the project.

Appendix C: Workscopes for U.S. University-led Integrated Research Project (IRP) R&D

PROGRAM DIRECTED: REACTOR CONCEPTS

INFRASTRUCTURE TO SUPPORT MOLTEN SALT REACTOR COMPONENTS (IRP-RC-1)
(FEDERAL POC –BRIAN ROBINSON & TECHNICAL POC – LOU QUALLS)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $5,000,000)

The Department of Energy is interested in working with universities to increase the amount of domestic research infrastructure available for the study of Molten Salt Reactors (MSR). Specifically, the objective is to develop and enhance domestic university capabilities to generate high-quality data, in coordination with the DOE MSR Campaign and MSR developers, which could be useful in the development and licensing of MSRs.

MSRs differ from current commercial plants in their fundamental design features, which leads to new technological challenges but also allows designers to take advantage of additional passive safety features and inherent protections. MSR component development and analysis as well as innovative engineering techniques for operations and reliability are sought to increase levels of safety and robustness, present new functionalities, and improve system performance.

Teams of universities, industry, and national laboratories are expected to provide a broad range of expertise and experience to the proposal. However, the emphasis on proposed efforts should include the establishment of new or enhanced research infrastructure at universities to broaden the base capability, to provide high-quality data for model validation or material property performance and prepare students to enter the emerging advanced reactor technical field. The development and/or expansion of university, industry and national laboratories irradiation facilities is strongly encouraged. Infrastructure support could include but are not limited to salt production, characterization and property measurement, and isotope production and isolation.

To ensure proposed infrastructure efforts complement existing research, specific examples are provided below. In addition to these examples, other proposals enhancing the domestic MSR research infrastructure are welcome.

EXPERIMENTAL VALIDATION OF THERMAL HYDRAULIC SIMULATIONS

Experimental Validation of Molten Salt Reactor Simulations should be focused on providing high quality data for the validation of system and computational fluid dynamics models of MSR (liquid fueled or pebble bed) phenomena. Multiple phenomena have been identified as relevant to core safety and performance but for which insufficient data exist for validating models and codes. Proposals should include: core heat transfer, plenum-to-plenum heat transfer by natural circulation, heated two-component stratified flow in the outlet plenum, bypass flow between fuel or reflector blocks, and performance of reactor cavity cooling systems in cooling the reactor guard vessel (ex-vessel heat transfer).

Validation of codes that capture these phenomena requires the coordinated completion of a number of fundamental, separate effects tests (SET), mixed effects tests (MET), such as combined mass flow and heat transfer, and integral tests, all properly scaled to reproduce the thermal fluid conditions bounding MSRs under nominal and accident scenarios.

Applications are sought that will fill the gaps in the data needed MSR code validation with appropriately scaled fundamental, SET, or MET experiments that complement those that have been, or can be, conducted at suitable, existing integral facilities. Investigators who wish to propose new experiments using one or more of DOE's existing facilities are strongly urged to coordinate with the Principal Investigators at those facilities before submitting the final application to obtain guidance on costs, schedule, and quality assurance.

PROGRAM DIRECTED: REACTOR CONCEPTS

ADVANCED HEAT EXCHANGERS

Molten salt reactor systems may benefit from advanced heat exchangers for improved cost and efficiency. Molten Salt Reactors (MSRs), which have a reactor output temperature (≈700°C) are good candidates for inclusion of advanced heat exchanger technology. Secondary heat exchangers would require accommodating the large differences in pressure between the low-pressure secondary salt coolant and the high-pressure working fluid in the power conversion system.

High efficiency heat exchangers have complex passageways that are a complicating factor for performing the rigorous stress analysis required to assess elevated temperature cyclic life under combined pressure and thermal gradient induced stresses. Other complicating factors are the number of structural features to be represented and stress concentrations at the corners of the flow channels. Because of this complexity, it is difficult to apply the normal stress classification process of the ASME Code. However, there are recently developed methodologies based on analysis approaches using an elastic-perfectly plastic (EPP) model to limit various stress measures and strain accumulation.

EXPERIMENTAL DATA FOR FISSION PRODUCT RETENTION, DIFFUSION AND TRANSPORT PROPERTIES

The objective is to study the release and transport behavior of radionuclides (gaseous, mists, foams) in liquid-fueled molten salt reactors under representative irradiation conditions. Separate effects testing on transport mechanisms, thermomechanical or thermophysical property influence, and primary structural system interaction that are related or coupled to modeling efforts are encouraged. However, the emphasis of this effort should be to develop data necessary to model radionuclide release from fuel salt surfaces and subsequent transport during both normal and accident conditions, including the effects of direct fission in the salt. Proposals are sought to provide separate effects fission product data for temperature and temporal dependent performance modelling. The use of university reactors to execute irradiations is strongly encouraged.

TARGETED IRRADIATIONS OF CORE INTERNAL AND BOUNDARY MATERIALS

The objective is to understand radiation damage effects (swelling, embrittlement, segregation, etc.) on advanced structural materials for representative molten salt reactors and also for candidate non-metal reactor core structural material, such as graphite or silicon carbide. Proposals are sought for irradiation and post-irradiation examination of MSR candidate core boundary and internal materials to provide data for validation data for structural performance, fabricability, and/or surface degradation. The use of university reactors to execute irradiations is strongly encouraged.

PROGRAM DIRECTED: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

NEAMS THERMAL HYDRAULICS AND FUELS (IRP-NEAMS-1)
(FEDERAL POC – DAVE HENDERSON & TECHNICAL POC – CHRIS STANEK)

Given the urgency to accelerate development and deployment of DOE-NE mod-sim capabilities to meet industry and NRC needs, a specific goal of this particular call is to recruit a team of university researchers to participate within the program, working closely with national laboratory colleagues in support the overall programmatic effort. The applied R&D performed by the program requires highly specialized expertise that is resident within universities and when combined with existing national laboratory efforts can importantly increase momentum of the development of DOE-NE modeling and simulation tools, which are those to be exclusively developed and utilized. In addition to technical contributions, it is further expected that university partners will assist with the definition of programmatic priorities and deliverables as well as potentially serving in program leadership positions. When appropriate, university partners are also expected to serve as representatives of the program representatives, much like their national laboratory counterparts, presenting technical progress at meetings, conferences, etc. Proposals should identify a university team, preferably a consortium of two or more universities, to maximize areas and depth of expertise, as well as mechanisms to solicit input from, prioritize research in concert with, and share results with the nuclear industry and the NEAMS program.

The above IRP description pertains to proposals sought in two topical areas: (1) university component of the existing Center of Excellence for Thermal-Fluids Applications in Nuclear Energy, and (2) university component in the NEAMS Fuel Performance Area.

IRP-NEAMS-1.1: THERMAL-FLUIDS APPLICATIONS IN NUCLEAR ENERGY (UP TO 3 YEARS AND $3,000,000)

In 2018, the Nuclear Energy Advanced Modeling and Simulation program initiated a Center of Excellence for Thermal-Fluids Applications in Nuclear Energy (https://neams.inl.gov/centerofexcellence). A primary goal of the Center of Excellence concept was to create a clear “front-door” to stakeholders (primarily industry and NRC) for access to DOE-NE thermal hydraulics capabilities and in particular modeling and simulation in order to perform research on novel solution strategies for historically challenging flow issues that still plague the current fleet of deployed Light Water Reactor (LWR) nuclear reactors as well as predicting various fluid flow and fluid related issues with advanced reactor technologies.

The Center of Excellence strives to advance a thermal-fluids research and development approach that synergistically combines three natural, though overlapping, length and time scales in a hierarchal multi-scale approach to avoid the temptation and pitfalls of attempting to develop a single “solve all” algorithm for physical fluid flow problems that will span 9 orders of magnitude in spatial and temporal scales. Key components of the early stage of the Center of Excellence included (1) improving the coherence of the hierarchal multi-scale thermal hydraulics modeling and simulation approach, and (2) communicating modeling and simulation capabilities through workshops and establishing pathways to collaboration with stakeholders. Now, given the early success of the Center of Excellence for Thermal-Fluids Applications in Nuclear Energy we are interested in expanding upon it by supporting university R&D.

We seek proposals to establish a university complement to the lab-led Center of Excellence that will leverage recent advancements to develop a CFD framework for the simulation of single-phase flow with the primary goal of facilitating the design and assessment of advanced reactors. It must be coupled with the fuel performance code BISON to deliver coupled CFD-fuel performance simulations, which are likely to play an increasingly larger role for future advanced fuel forms. Finally, the proposals should identify plans to integrate the results obtained with these advanced models in the overall multiscale approach of the center of excellence, supporting the creation of reduced order models or novel closures for use in systems codes (e.g., SAM, RELAP-7).

PROGRAM DIRECTED: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

IRP-NEAMS-1.2: MULTISCALE NUCLEAR FUEL PERFORMANCE UP TO 3 YEARS AND $3,000,000)

Given the importance of nuclear fuel performance in many of the non-LWR designs, we seek proposals for a university consortium that contributes to the advancement of the BISON nuclear fuel performance code for non-LWR advanced reactors. In particular, in order to accelerate the development of BISON for the simulation of particle fuel, proposals are sought that focus on the thermomechanical behavior of particle fuel, including development of materials and behavioral models (including UCO and UC fuel kernels) and where appropriate coupling to thermal hydraulics and neutronics. Proposals should account for research already under way, such as that under existing CINR/NEUP awards as well as the Scientific Discovery through Advanced Computation – SciDAC (https://collab.cels.anl.gov/display/FissionGasSciDAC2). It is further expected that the university team work in concert with national laboratory researchers to establish priorities and perform technical research. https://neams.inl.gov/SitePages/Fuels%20Product%20Line.aspx